Appendix E
New Research Reactor Operations

E.1 INTRODUCTION

A preconceptual design of a new research reactor was developed to meet the U.S. Department of Energy's (DOE) missions of (1) producing medical and industrial radioisotopes, (2) producing plutonium-238 (minimum net annual production of 5 kilograms [11 pounds]), and (3) supporting nuclear energy research and development. A design goal of this new research reactor is that it should use only low-enriched uranium with an enrichment of less than 20 percent uranium-235, consistent with U.S. nonproliferation policy. This preconceptual design includes the basic elements of the research reactor facility, which are sufficient to support this programmatic environmental impact statement, but does not include design details (i.e., system and layout drawings, bill of materials, electrical and piping routing, etc.) commensurate with a complete preliminary reactor design. The reactor design uses an existing proven and licensed low-enrichment fuel type in conjunction with a reactor system that features numerous inherent safety features, such as:

- A large low-pressure, low-temperature coolant inventory around the core to keep the core covered under all accident conditions, absorb heat, and filter any released radioisotopes
- A large prompt negative fuel temperature coefficient of reactivity to mitigate any accidental reactivity insertion event
- Extensive research, development, and operational experience in conjunction with a robust nuclear fuel design
- Minimal reliance on the operation of any active system or component for safe shutdown and accident response

Although significant additional work would be required to develop a detailed preliminary design of this research reactor, the preconceptual design provides the basis for evaluating the environmental impacts and cost of this alternative.

E.2 NEW RESEARCH REACTOR GENERAL DESCRIPTION

The design of the new research reactor is based on current research reactor designs, which have been approved by both the U.S. Nuclear Regulatory Commission (NRC) and the International Atomic Energy Agency (IAEA), as well as nuclear regulatory authorities of many nations. Most low-enriched uranium operating research reactors use one of two types of nuclear fuel: (1) uranium-aluminum matrix, or (2) uranium-zirconium-hydride (UZrH) with either a light or heavy water neutron moderator. In addition, most research reactors are based on either a pool or tank enclosure for the reactor core. Research reactor designs are based on several key factors including, but not limited to: (1) mission, (2) enrichment limits, (3) required neutron flux, (4) thermal limits, (5) irradiation volume, (6) safety, (7) operations, and (8) cost. All these factors were considered in the preconceptual design of this new research reactor.

Scoping reactor core physics calculations were performed using the SCALE-4.4 (ORNL 1998a) computer code package to evaluate three different low-enriched uranium nuclear fuel designs: (1) ternary uranium-zirconia-calcium oxide clad in stainless steel 304 similar to that used in the Power Burst Facility reactor at the Idaho National Engineering and Environmental Laboratory (INEEL) (ANC 1971), (2) uranium-aluminum-silicide alloy clad in aluminum (NRC 1988), used in many research reactors and (3) UZrH alloy clad in Incoloy-800, which is known as TRIGA (training, research, isotopes General Atomics) fuel. All core physics analyses assumed a light water moderator instead of the alternative heavy water moderator because of the significant additional cost related to the use of heavy water. Based on this scoping analysis, coupled with the desired mission of this reactor, current nuclear fuel manufacturing capability, and safety considerations, a TRIGA fuel design was selected for the new research reactor. The principal distinguishing features of the TRIGA fuel are its proven safety performance during power pulsing and its demonstrated long-term irradiation integrity.

To produce the desired quantity of plutonium-238 along with medical and industrial radioisotopes concurrently with nuclear research and development, it was determined that a reactor core power of 50 megawatts-thermal would be adequate. Although it was not evaluated for environmental impacts, the core and reactor systems were designed to accommodate a power level of up to 100 megawatts. At the 50 megawatts-thermal power level, the core requires an active cooling system with forced coolant flow to maintain the fuel below its material thermal limits. The new research reactor cooling system design uses a tank within a pool, which is connected to primary coolant circulating pumps, heat exchangers, and an ultimate heat sink consisting of two cooling towers. The pool is housed in a reactor building which also encloses the pumps, heat exchangers, secondary systems, and spent nuclear fuel storage pool. The spent nuclear fuel storage pool, sized to store the reactor core's discharged spent nuclear fuel for its entire 35-year lifetime, can be hydraulically connected to the reactor core pool for refueling and emergency reflooding. The ultimate heat sink cooling towers, air exhaust stack, and emergency diesel generators are located outside the reactor building.

E.3 NEW RESEARCH REACTOR FUEL AND CORE DESIGN

As discussed in Section E.2, TRIGA fuel was selected for the new research reactor core. TRIGA fuel has been used in research reactors since 1958 with over 50 TRIGA reactors currently operating worldwide, including 19 operating U.S. TRIGA reactors that are licensed by the NRC (NRC 1999), at licensed steady-state power levels of 0.02 to 16 megawatts-thermal and power pulsing capabilities of up to 22,000 megawatts-thermal (GA 2000; Simnad 1980; UC-Davis 1999). This fuel design has demonstrated ability to provide high burnup cladding integrity as well as reliable performance up to actual burnups of 75 atom percent uranium-235 (Simnad 1980). The TRIGA fuel, because of its unique composition of hydrogen moderator intimately mixed into the fuel itself, has a large negative fuel temperature reactivity coefficient, which shuts the reactor down during any power excursions or reactivity-induced transients.

E.3.1 Nuclear Fuel Design

The new research reactor nuclear fuel design is based on an extension of current licensed low-enriched uranium TRIGA fuel designs for 10- to 16-megawatts-thermal reactors. A comparison of the current high-power low-enriched uranium TRIGA fuel design and the new research reactor low-enriched TRIGA fuel design (IAEA 1992) is presented in **Table E–1**.

Table E–1 Comparison of New Research Reactor Fuel Design to Current Low-Enriched Uranium TRIGA Fuel Design

Nuclear Fuel Design Parameter	Current Low-Enriched Uranium TRIGA Fuel Value	New Research Reactor TRIGA Fuel Value
Fuel assembly rod configuration	Square 4 rods by 4 rods	Square 8 rods by 8 rods
Fuel assembly shroud outside dimension	7.572 centimeters (2.981 inches) by 7.963 centimeters (3.135 inches)	16.52 centimeters (6.5 inches) by 16.52 centimeters (6.5 inches)
Fuel rod center-to-center pitch	1.634 centimeters (0.643 inch)	2 centimeters (0.787 inch)
Fuel rod cladding outside diameter	1.377 centimeters (0.542 inch)	1.377 centimeters (0.542 inch)
Cladding material	Incoloy-800	Incoloy-800
Cladding thickness	0.041 centimeter (0.016 inch)	0.036 centimeter (0.014 inch)
Fuel-to-cladding radial gap	0.0022 centimeter (0.0009 inch)	0.0025 centimeter (0.001 inch)
Fuel rod gap and gas plenum backfill gas and fill gas pressure	Helium 0.0103 megapascal (1.5 pounds per square inch absolute)	Helium 0.0103 megapascal (1.5 pounds per square inch absolute)
Fuel pellet outer diameter	1.295 centimeters (0.510 inch)	1.295 centimeters (0.510 inch)
Fuel pellet height	13.97 centimeters (5.5 inches)	13.97 centimeters (5.5 inches)
Fuel pellet composition	$UZrH_{1.6}$-Er	$UZrH_{1.6}$-Er
Fuel pellet uranium weight fraction	45 percent	45 percent
Uranium-235 enrichment	19.95 percent	19.7 percent
Hydrogen-to-zirconium ratio	1.6	1.6
Fuel pellet erbium weight fraction	0.8 percent	0.8 percent
Fuel rod active fuel length	55.88 centimeters (22.0 inches)	153.7 centimeters (60.5 inches)
Mass of uranium per fuel rod	274 grams (0.604 pound)	754 grams (1.660 pounds)
Mass of uranium-235 per fuel rod	54.8 grams (0.121 pound)	148.4 grams (0.327 pound)
Mass of uranium per "all fuel rod" fuel assembly	4.38 kilograms (9.64 pounds)	48.3 kilograms (106.2 pounds)
Total fuel rod length	76.2 centimeters (30 inches)	176 centimeters (69.3 inches)

Key: $UZrH_{1.6}$-Er, uranium-zirconium-hydride with erbium.

As presented in Table E–1, the new research reactor fuel design is identical to current low-enriched TRIGA fuel for higher power cores except that the new reactor fuel has a larger assembly configuration array (i.e., 8 by 8 versus 4 by 4) and a longer active fuel length (153.7 centimeters versus 55.88 centimeters). The larger array and longer length were selected to meet the plutonium-238 production requirements and maintain high safety factors with respect to fuel thermal performance.

E.3.2 Nuclear Core Design

Along with fuel rods, the core is designed to contain a number of plutonium-238, medical radioisotope, and industrial radioisotope production target rods. These target rods would occupy positions in a fuel assembly where a fuel rod would otherwise exist. Each of these positions would have an Incoloy-800 alloy guide tube with the same dimensions as the fuel rod cladding. The target rods would be inserted into these guide tubes for their design irradiation time period. In addition, some fuel rod positions in core fuel assemblies would be replaced with similar guide tubes to accommodate Incoloy-800-clad boron carbide control rods. Boron carbide is a proven, accepted, and widely used neutron absorber for control rods. **Figure E–1** presents a representative illustration of the fuel rod, neptunium-237 target rod, medical and industrial radioisotope target rod, and control rod. **Figure E–2** shows a cross section of each type of fuel assembly in the core.

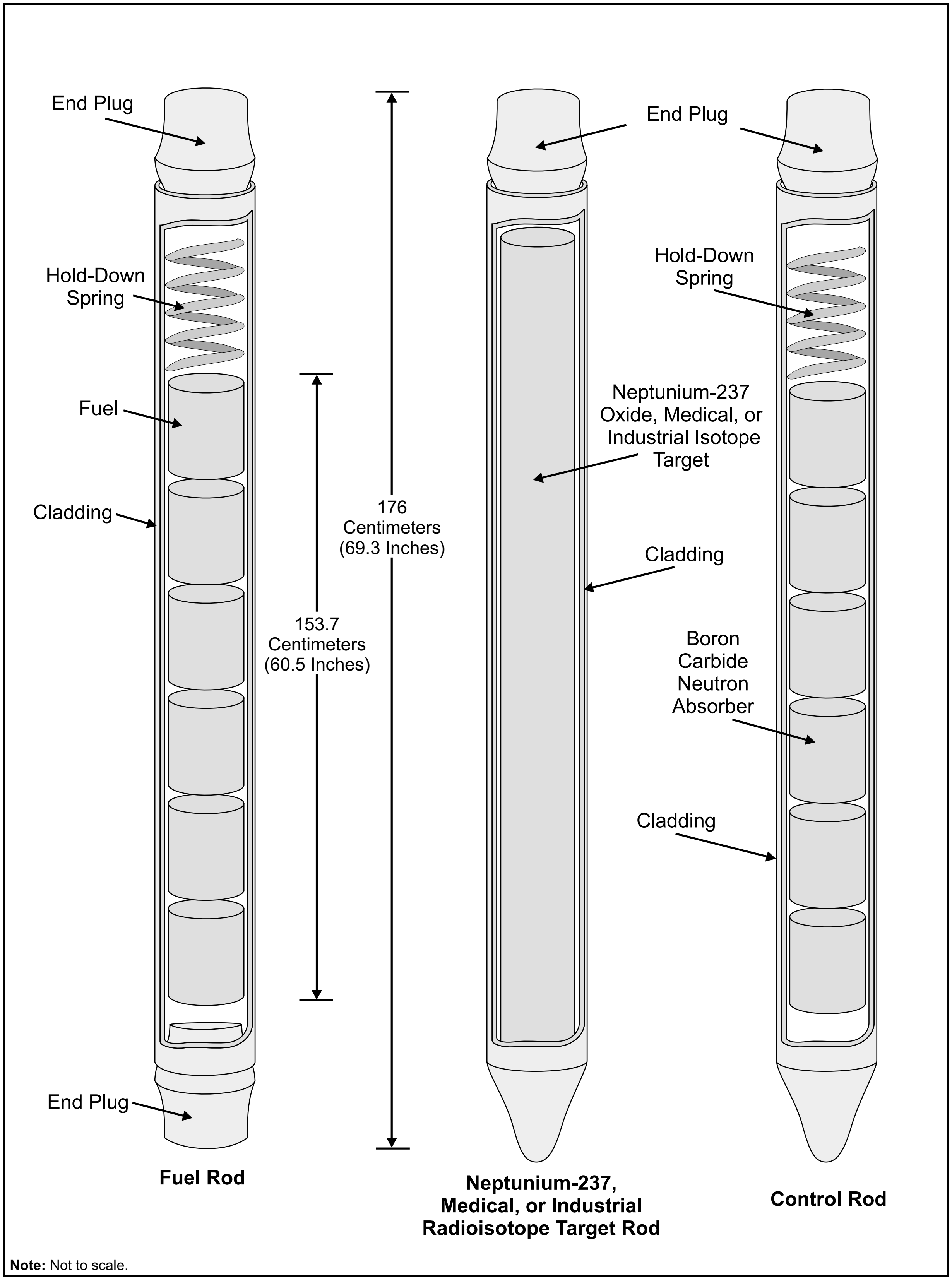

Figure E–1 Representative Illustration of Fuel Rod; Neptunium-237, Medical, or Industrial Radioisotope Target Rod; and Control Rod

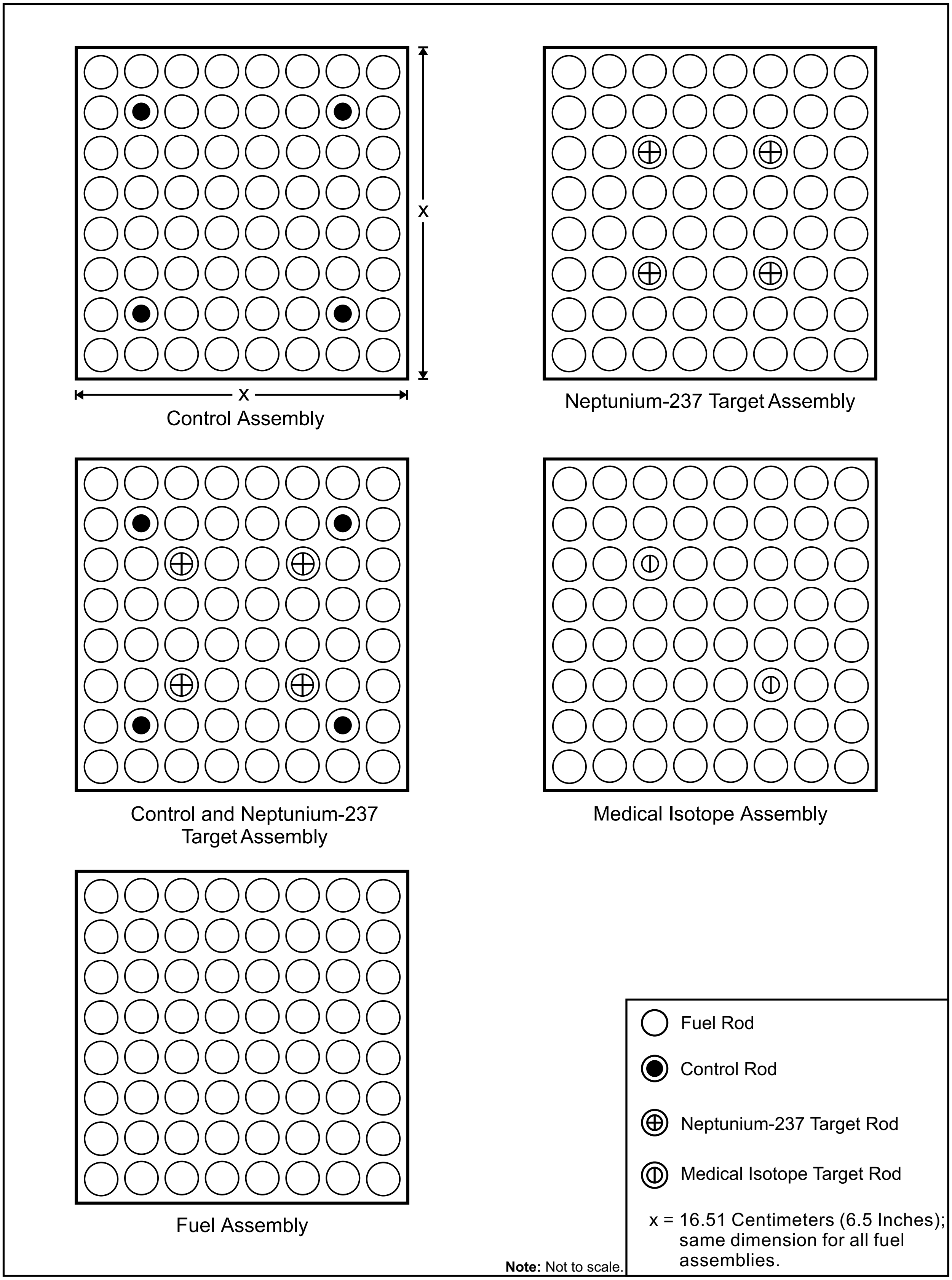

Figure E–2 Cross Section of Fuel Assemblies in the Core

The new research reactor core design consists of 68 fuel assemblies, each of which is enclosed in a square aluminum shroud for structural support and coolant flow control. Key design features of the core are presented in **Table E–2**.

Table E–2 Key Design Features of the New Research Reactor Core

Core Design Parameter	Value
Number of fuel assemblies	68
Core thermal power	50 megawatts
Average fuel assembly power	0.74 megawatt
Number of neptunium-237 target rod assemblies	48
Number of neptunium-237 target rods per assembly and in core	4 per assembly, 192 in core
Number of medical and industrial radioisotope target rod assemblies	8
Number of medical and industrial radioisotope target rods per assembly and in core	2 per assembly, 16 in core
Available radioisotope production volume	42.1 liters (1.5 cubic feet)
Number of control rod assemblies	16
Number of control rods per assembly and in core	4 per assembly, 64 in core
Total number of fuel rods in core	4,080
Core active height	153.7 centimeters (60.5 inches)
Core diameter	166.5 centimeters (65.5 inches)
Core radial reflector material and minimum thickness	Beryllium, 3.0 centimeters (1.18 inches)
Core uranium mass	3.1 metric tons of uranium (6,820 pounds)
Core uranium-235 mass	611 kilograms (1,344 pounds)
Minimum core life at 80 percent capacity factor	10 years

The core design described in Table E–2 also includes eight rabbit tubes for short irradiation-time production of medical or industrial radioisotopes and nuclear research and development. These rabbit tubes are located outside the fuel region of the core, but still within an area with a relatively high neutron flux. A cross-sectional view of the new research reactor core showing the layout of fuel assemblies, target rod assemblies, control rod assemblies, reflector, and rabbit tubes is presented in **Figure E–3**.

E.4 NUCLEAR FUEL THERMAL PERFORMANCE

The nuclear fuel design, based on core physics analyses, also was evaluated to determine if its thermal performance would meet the relevant thermal limits for TRIGA fuel. The steady-state thermal performance of the new research reactor nuclear fuel was analyzed to evaluate three indicators: (1) peak fuel pellet centerline temperature, (2) critical heat flux ratio (sometimes denoted as departure from nucleate boiling ratio), and (3) fuel rod internal volume gas pressure and associated cladding hoop stresses.

Peak fuel pellet temperature is an important variable for TRIGA fuel because, at elevated temperatures, hydrogen within the UZrH fuel matrix is released as a gas and can cause excessive pressure that may result in cladding rupture. A fuel temperature limit of 650 °C (1,202 °F) precludes excessive hydrogen gas release and pressure (Simnad 1980).

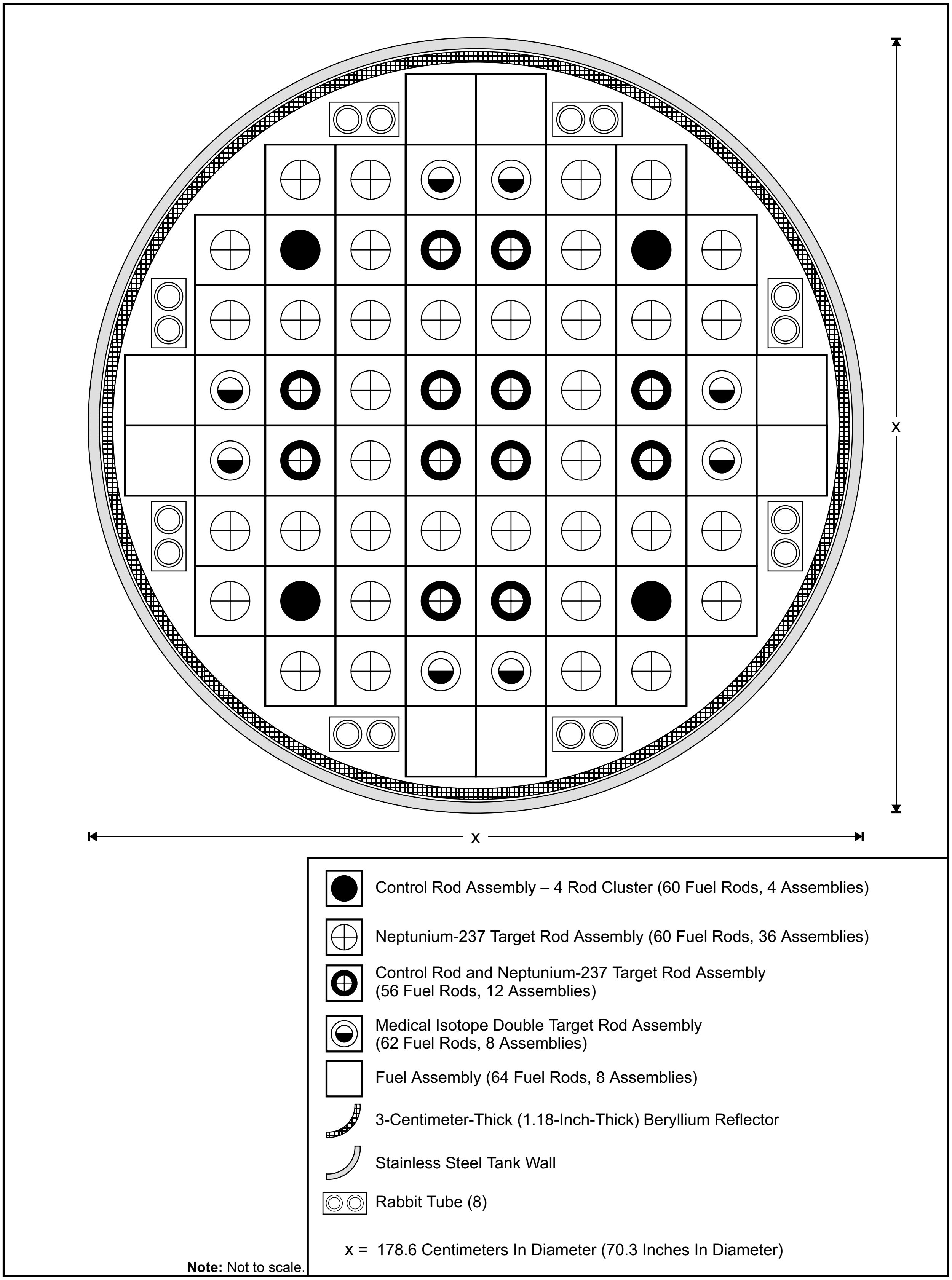

Figure E–3 Cross-Sectional View of Research Reactor Core

The critical heat flux ratio is a measure of the nature of the heat transfer from the fuel rod cladding surface to the coolant water flowing past it. As the cladding surface heat flux increases, heat transfer to the coolant increases until critical heat flux is reached. Beyond this critical heat flux, further increases in surface heat flux will not result in greater heat transfer to the coolant, thus causing the cladding and fuel temperature to rise rapidly. By maintaining the fuel cladding surface conditions below critical heat flux, a safe heat transfer regime exists between the fuel and the coolant. The critical heat flux ratio is the calculated critical heat flux, for a given set of peak core thermal-hydraulic conditions, to the actual cladding maximum surface heat flux. A critical heat flux ratio greater than 1 is an indicator of an acceptable thermal design. Typically, due to the statistical uncertainty and experimental conditions associated with maximum core heat flux and critical heat flux correlations, a ratio of greater than 1.2 is desirable.

The calculated peak fuel rod internal gas pressure and associated cladding hoop stress are an indication of the structural integrity of the cladding, which constitutes an important fission product confinement barrier. As the fuel is irradiated in the reactor core, gaseous fission products are produced. Some of these fission products can escape from the fuel pellets and collect in the gap between the fuel and cladding as well as the plenum volume above the fuel pellet stack inside the fuel rod. An increase in the presence of gaseous fission products inside the fuel rod volume results in a higher rod internal pressure and concomitant higher cladding hoop stress over the lifetime of the fuel. It should be noted that the UZrH fuel pellet material has been shown to retain most of its fission products at operating temperatures (Simnad 1980).

The radial temperature distribution of an axial segment in the average and peak fuel rod was calculated using the one-dimensional solution to the conservation of energy equation for a cylindrical geometry. Appropriate values for coolant and fuel rod material thermal-physical properties were used in this analysis. Bounding assumptions were made for boundary conditions and fuel rod peaking factor. **Table E–3** presents the thermal parameters and important results relevant to the analysis of fuel rod temperatures.

Table E–3 Fuel Rod Temperature Distribution Analysis Parameters for Steady-State Operation

Fuel Rod Temperature Parameter	Value
Core power density	14.9 kilowatts per liter
Average fuel rod linear heat generation rate	79.7 watts per centimeter (8,291 Btu/hr/ft)
Average fuel rod surface heat flux	18.43 watts/cm^2 (58,418 Btu/hr/ft^2)
Maximum fuel rod peaking factor	2.25
Average core coolant flow rate	1.1 meters per second (3.6 feet per second)
Core hottest channel inlet coolant temperature	46.1 °C (115 °F)
Average fuel channel axial coolant temperature rise	10.9 °C (19.7 °F)
Hottest fuel channel axial coolant temperature rise	24.4 °C (44 °F)
Cladding surface temperature Average fuel rod Peak fuel rod	 78.9 °C (174 °F) 113.1 °C (235.6 °F)
Fuel pellet centerline temperature Average fuel rod Peak fuel rod	 160 °C (320 °F) 285.7 °C (546.3 °F)
Fuel pellet normal operation temperature limit	650 °C (1,202 °F)
Minimum critical heat flux ratio Average fuel rod Peak fuel rod	 2.91 1.30
End-of-core-life fuel rod internal volume pressure Average fuel rod Peak fuel rod	 0.146 MPa (21.2 psia) 0.170 MPa (24.6 psia)
End-of-core-life fuel rod cladding hoop stress (external pressure = 0.101 Mpa [14.7 psia]) Average fuel rod Peak fuel rod	 0.855 MPa (124 psi) 1.303 MPa (189 psi)
Cladding yield strength	251.5 MPa (36,500 psi)
Cladding ultimate strength	900 MPa (130,500 psi)

Key: Btu/hr/ft^2, British thermal units per hour per square foot; MPa, megapascal; psi, pounds per square inch; psia, pounds per square inch absolute; watts/cm^2, watts per square centimeter.

Table E–3 shows that the maximum calculated fuel pellet centerline temperature, minimum critical heat flux ratio, and maximum fuel rod cladding stress are well within their relevant limits. This analysis demonstrates that the new research reactor fuel and core designs meet basic thermal design criteria with ample safety margins for steady-state operation at a 50 megawatt-thermal core power level.

E.5 NUCLEAR CORE PHYSICS PERFORMANCE

Nuclear core physics calculations were performed with three state-of-the-art digital computer codes and/or code packages: SCALE-4.4 (ORNL 1998a), WIMSDB5 (ORNL 1998b), and MCNP4B2 (Briesmeister 1999).

SCALE-4.4 uses discrete ordinate deterministic methods, one-dimensional unit cell geometry, transport theory, and point depletion to calculate neutron flux, k-infinity, and fission product inventory. K-infinity is a measure of neutron multiplication factor for a system of infinite size, which is an indication of reactivity to meet the mission during a core cycle of operation. A minimum value of about 1.1 for k-infinity is required to attain a critical state and produce neutrons for radioisotope production. SCALE-4.4 was used with a 44-energy-group cross-section database to calculate fission products and as an independent validation of k-infinity and neutron flux. WIMSD Version 5B uses a one-dimensional transport theory deterministic method with a lattice representation of the fuel assembly that accounts for fuel rod and assembly geometry and a 69-energy-group cross-section database to calculate neutron flux, k-infinity, and plutonium-238 production rate. WIMSD Version 5B was used to calculate plutonium-238 production rate and neutron flux and as an independent validation of the values of k-infinity. MCNP4B2 uses a three-dimensional monte carlo stochastic transport particle simulation of the full core geometry with continuous, point-wise neutron cross-section data to calculate the k-infinity, spatial neutron flux distribution in the core, and the neutron flux energy spectrum. MCNP4B2 was used to calculate peak and average core neutron flux, in terms of both energy and spatial distribution, and k-infinity.

The three computer codes predicted beginning of core life values of k-infinity to within about 1 percent of each other. Comparisons of calculated core neutron flux were also close among the different computer codes. In addition, a review of maximum thermal neutron flux for operating TRIGA research reactors with a core power from 1 to 14 megawatts showed values of 1×10^{13} to 3×10^{14} neutrons per square centimeter per second (IAEA 1989; ANL 2000). This range of neutron flux compares well with the calculated maximum value of 6×10^{13} neutrons per square centimeter per second for this new 50-megawatt TRIGA-based research reactor. Key core physics results are presented in **Table E-4.**

Table E–4 Key Core Physics Performance Parameters

Core Physics Parameter	Value
Beginning-of-life k-infinity	1.5
End-of-life k-infinity	1.1
Plutonium-238 production after 300-day operation	5.3 kilograms (11.7 pounds)
Average core thermal (less than 0.625 electron volt) neutron flux	2.5×10^{13} neutrons/cm^2/sec
Peak core thermal neutron flux	6×10^{13} neutrons/cm^2/sec
End-of-life fuel assembly Average burnup Maximum burnup	 2,292 megawatt-days 5,157 megawatt-days
Core average end-of-life uranium-235 atom burnup	34 percent

Key: neutrons/cm^2/sec, neutrons per square centimeter per second.

Table E–4 shows that the core is designed for a minimum useful life of approximately 10 years and would produce greater than 5 kilograms (11 pounds) of plutonium-238 during a 300-day operating period. It should be noted that limited core physics calculations were performed as part of this new research reactor preconceptual design. Detailed spatial full core depletion calculations were not performed for this reactor. The numerical core physics performance and plutonium-238 production values presented in Table E–4 are based on conservative calculations. The neptunium-237 target design was based on a limitless supply of neptunium-237. A significant reduction (by a factor of four to eight) in the required neptunium-237 target mass for the same annual plutonium-238 production rate can be achieved by: (1) optimizing target design with annular pellets, (2) inclusion of moderator materials mixed in the target, (3) optimizing fuel geometry, and (4) increasing core power.

E.6 PRIMARY COOLANT SYSTEM DESIGN

The major components of the primary coolant system design for the new research reactor are (1) a reactor core pool, (2) two primary coolant pumps, (3) two primary coolant heat exchangers, (4) shutdown pumps, and (5) piping between the pool, pumps, and heat exchangers. This system is designed to remove core thermal power during normal operation and core decay heat after reactor shutdown while maintaining the fuel below its thermal limits. A schematic of the primary coolant system is presented in **Figure E–4.**

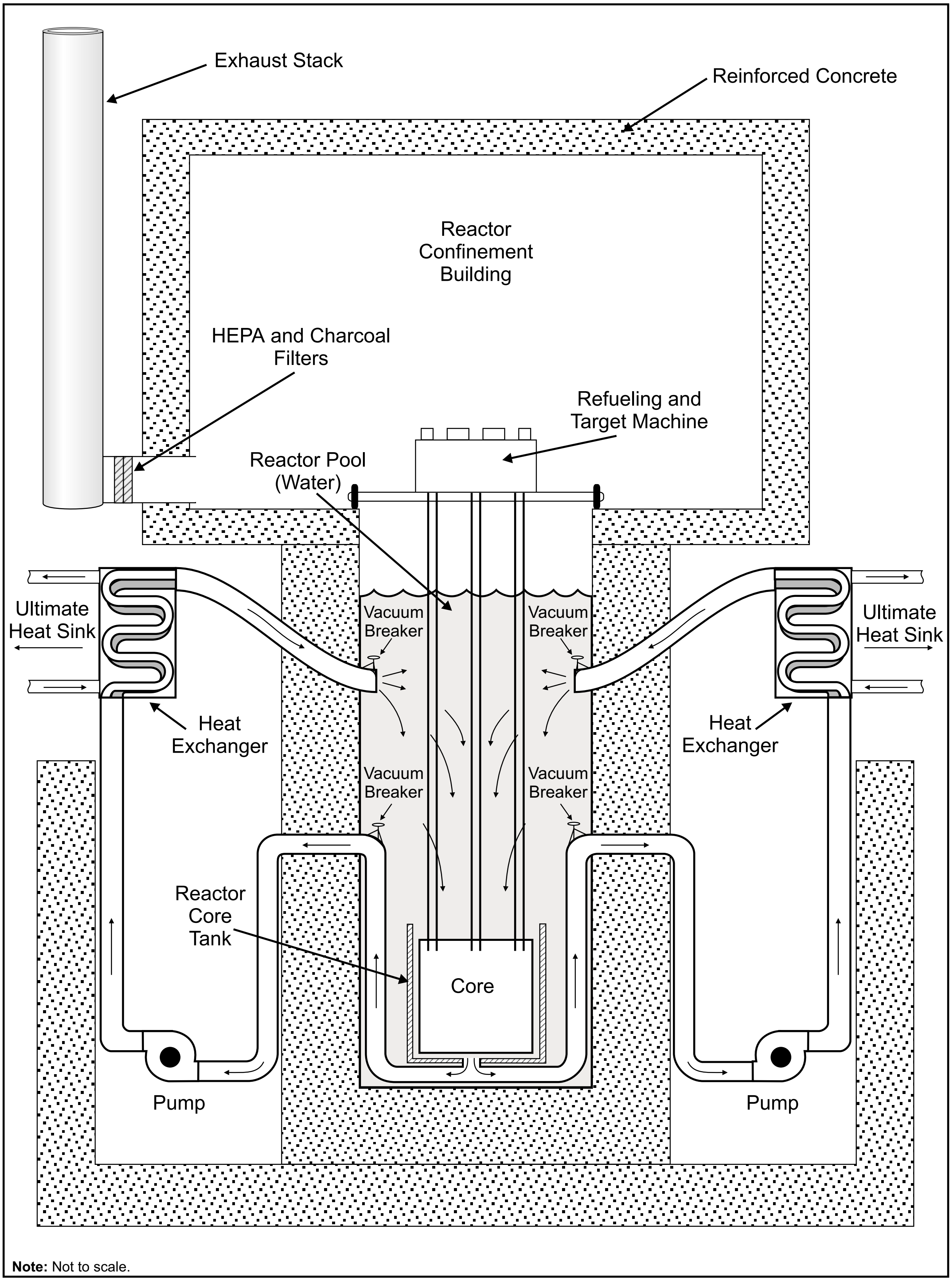

Figure E–4 Schematic of Primary Coolant System

Coolant flow would be drawn down through the core by the primary coolant pump suction pipelines, which would be connected to the bottom of the tank enclosing the core inside the pool. The side and bottom of the cylindrical tank would be sealed, but the top of the tank would be open to the pool. The forced coolant downflow through the core, as opposed to upflow, is designed to mitigate radiological consequences from the production of nitrogen-16, which is created from fast neutron reaction with the oxygen in the water molecule. Nitrogen-16 is a radioisotope which emits high-energy gamma (6 million electron volts per disintegration) radiation, but has a decay half-life of approximately 7 seconds. Therefore, the core coolant downflow allows this nitrogen-16, produced in the core coolant region, to decay before it returns to the reactor core pool. The primary coolant system loop (i.e., core to pump to heat exchanger to pool) is designed to delay the coolant removed from the core by at least 21 seconds (i.e., three half-lives) prior to its return to the pool. Appropriate shielding would be designed for all primary coolant system components to protect workers from nitrogen-16 gamma radiation to allow access during power operation.

Another design feature of the primary coolant system intended to reduce nitrogen-16 radiation at the top of the reactor core pool is the submerged location of the return flow from the heat exchanger. This submerged location allows shielding of the water depth above it and further nitrogen-16 decay before it can diffuse to the top of the pool.

The primary coolant system design has several inherently safe design elements that preclude or mitigate postulated accidents. As illustrated in Figure E–4, the suction piping from the bottom of the core tank would be routed, within the pool, to an elevation of 3 meters (10 feet) above the top of the core before it exits the pool through the pool wall and down to the pump in an adjacent compartment. A vacuum breaker (antisyphon device) would be attached to the high point of this suction piping inside the pool to prevent draining the reactor pool during any postulated pipe break accident. The balance of the primary coolant system (i.e., pump discharge piping, heat exchanger, and heat exchanger outlet piping) all would be elevated above the top of the core so that any leak or rupture could not uncover the core. This primary coolant system configuration precludes any leak from draining the pool below the top of the core.

The reactor core pool would be fully lined with 304 stainless steel, which would be attached to reinforced concrete. The reinforced concrete walls and floor of the pool are designed to meet all design-basis earthquake loads in the United States outside of coastal California.

All primary coolant system components would be type 304 stainless steel and would be manufactured to meet current nuclear safety and quality assurance standards. The use of two 100 percent capacity pumps and heat exchangers would allow for normal core power operation in the event of the loss of one component. A low flow-rate shutdown pump would be included in the system to provide sufficient flow for core decay heat removal. Key primary coolant system design parameters are presented in **Table E–5.**

The design of the primary coolant system employs accepted and widely used nuclear power plant safety principles such as redundancy, single-failure-proof, nuclear quality assurance, and inherent natural phenomena (e.g., elevated piping).

Table E–5 Key Primary Coolant System Design Parameters

Design Parameter	Value
Reactor core pool water dimensions	7.9 meters (26 feet) long 6.7 meters (22 feet) wide 9.1 meters (30 feet) deep
Reactor core pool materials	304 stainless steel, 1.27-centimeter-thick (0.5-inch-thick) liner over reinforced concrete
Primary coolant system pump flow rate (100 percent capacity each)	1.262 cubic meters per second (20,000 gallons per minute)
Primary coolant system pump design head	61 meters (200 feet)
Primary coolant system heat exchanger design heat removal rate (100 percent capacity each)	50 megawatts (170×10^6 British thermal units per hour)
Primary coolant system heat exchanger design inlet temperature	51.7 °C (125 °F)
Primary coolant system heat exchanger design outlet temperature	40.6 °C (105 °F)
Primary coolant system shutdown pump flow rate	0.063 cubic meters per second (1,000 gallons per minute)
Primary coolant system shutdown pump design head	61 meters (200 feet)

E.7 Balance of Reactor Plant Systems

Due to the preconceptual design nature of the new research reactor, limited details have been developed for the balance of the reactor plant systems. Details were established where they were judged to significantly affect the determination of environmental impacts or cost.

The secondary cooling system design transfers the heat removed from the core by the primary coolant system to the environment. This system would consist of two 100-percent-capacity pumps and two 50-percent-capacity cooling towers with piping connecting them to each other and to the secondary side of the primary coolant system heat exchangers. Unlike the primary coolant system, the secondary coolant system is not designed or considered to be a nuclear-safety-related system because its failure would not challenge the safety of the reactor. Piping and cooling tower tubes would be constructed of carbon steel. To avoid causing fogging at the reactor building area, the cooling towers would be located about 122 meters (400 feet) from the reactor building.

The spent fuel pool is designed to the same standards and with the same materials as the reactor core pool. The spent fuel pool would be sized to store all the fuel expected to be discharged over the 35-year lifetime of the reactor. The spent fuel pool also is designed to accommodate shipping casks for transport of radioisotope target rods and the spent nuclear fuel after this reactor is shut down, decontaminated, and decommissioned. The reactor core pool can be hydraulically connected to the spent fuel pool using an isolable transfer canal for moving spent nuclear fuel from and reloading the core, as well as transferring isotope target rods from the core to the spent fuel pool. An important safety feature of this canal is its ability to flood the reactor core pool in the unlikely event of a coolant leak from the reactor core pool. The size of the spent fuel pool allows it to completely reflood the entire volume of the reactor core pool, using the transfer canal, while maintaining a sufficiently high level of water above the stored spent fuel to preclude high dose rates in the spent fuel pool area. Another unique safety feature of the spent fuel pool is the design of its fuel storage racks. These racks are designed to separate each fuel assembly from other fuel assemblies by approximately 23 centimeters (9 inches), which neutronically isolates each assembly, thereby assuring criticality safety without the use of soluble or fixed neutron absorbers. The space between fuel assemblies in the rack is covered with steel bars that prevent any accidental insertion of a fuel assembly. In addition, each storage position is recessed more than 23 centimeters (9 inches) below the top of the rack to avoid a dropped assembly neutronically interacting with another assembly. This simple storage rack design provides inherent criticality safety in a cost-effective design.

A spent fuel pool cooling system is incorporated into the facility design. This nuclear-safety-released system would be capable of maintaining the spent fuel pool temperature within acceptable limits under all modes of plant operation. This system would consist of two redundant small pumps, heat exchangers, and appropriate instrumentation. Spent nuclear fuel decay heat removed from the spent fuel pool would be transferred to the secondary cooling system.

A water makeup and purification system would be included in the reactor design for maintaining the water level in both the reactor core and spent fuel pools and removing any contamination. This system, consisting of a small pump, piping, and appropriate filters, resin beds, and makeup water tank, would also be used to maintain the chemistry of the water to within technical specification limits. Periodic monitoring of pool water quality would be used as an indicator of fuel failure or heat exchanger tube leakage.

The reactor instrumentation and control systems are expected to be similar to other higher-power TRIGA reactors and are nuclear safety related. Instrumentation would monitor important nuclear and thermal-hydraulic parameters with digital displays in the control room. Although TRIGA fuel has an inherently large negative temperature reactivity coefficient, a reactor trip system would scram (loss of power would cause the control rods to drop in the reactor core) the reactor on a number of redundant signals such as high power, low pool level, low coolant flow rate, and high core exit coolant temperature.

In the unlikely event of a loss of all offsite alternating current power, a nuclear safety-related emergency power system consisting of two redundant 1,506-kilowatt emergency diesel generators was included in the design. These emergency diesel generators would be nuclear safety grade and subject to periodic testing to ensure their reliability.

The reactor building heating, ventilating, and air conditioning system is designed to maintain the air temperature within the building within specified limits while removing contaminants and certain radioisotopes that may be present in the building air. This system would consist of an interconnected network of ducts, fans, chillers, heating coils, filters, and a 36.6-meter-high (120-foot-high) exhaust stack. High-efficiency particulate air (HEPA) and charcoal filters would remove a minimum of 99.9 percent of airborne particulates and 99 percent of airborne iodine. The exhaust portion of the heating, ventilating, and air conditioning system, which would perform the contaminant and radioisotope removal function, would be safety-related.

The reactor building complex and associated structures are schematically presented in **Figures E–5 and E–6.** The reactor building would consist of three sections, separated by a radiation shield wall: (1) reactor room, (2) system room, and (3) spent fuel pool. The reactor room would house the reactor core pool and provide confinement. The reactor system section would include: (1) primary coolant system pumps and a shutdown pump; (2) primary coolant system heat exchangers; (3) secondary coolant system pumps; (4) a water makeup and purification system; and (5) a 20-ton-capacity, nuclear-safety-related, single-failure-proof overhead crane. The spent fuel pool section would include: (1) a spent fuel pool; (2) a spent fuel pool makeup and purification system; (3) a pool transfer canal; (4) a loading dock; and (5) 20-ton-capacity, nuclear-safety-related, single-failure-proof overhead crane. The exterior walls and roof of the reactor building, consisting of reinforced concrete, are designed to withstand design-basis tornado missiles and seismic events for the entire United States outside of coastal California in accordance with current DOE and NRC regulations. The total footprint of the facility, including the reactor and control buildings, cooling towers, emergency diesel generators, exhaust stack, and ancillary structures, is estimated to be approximately 3,623 square meters

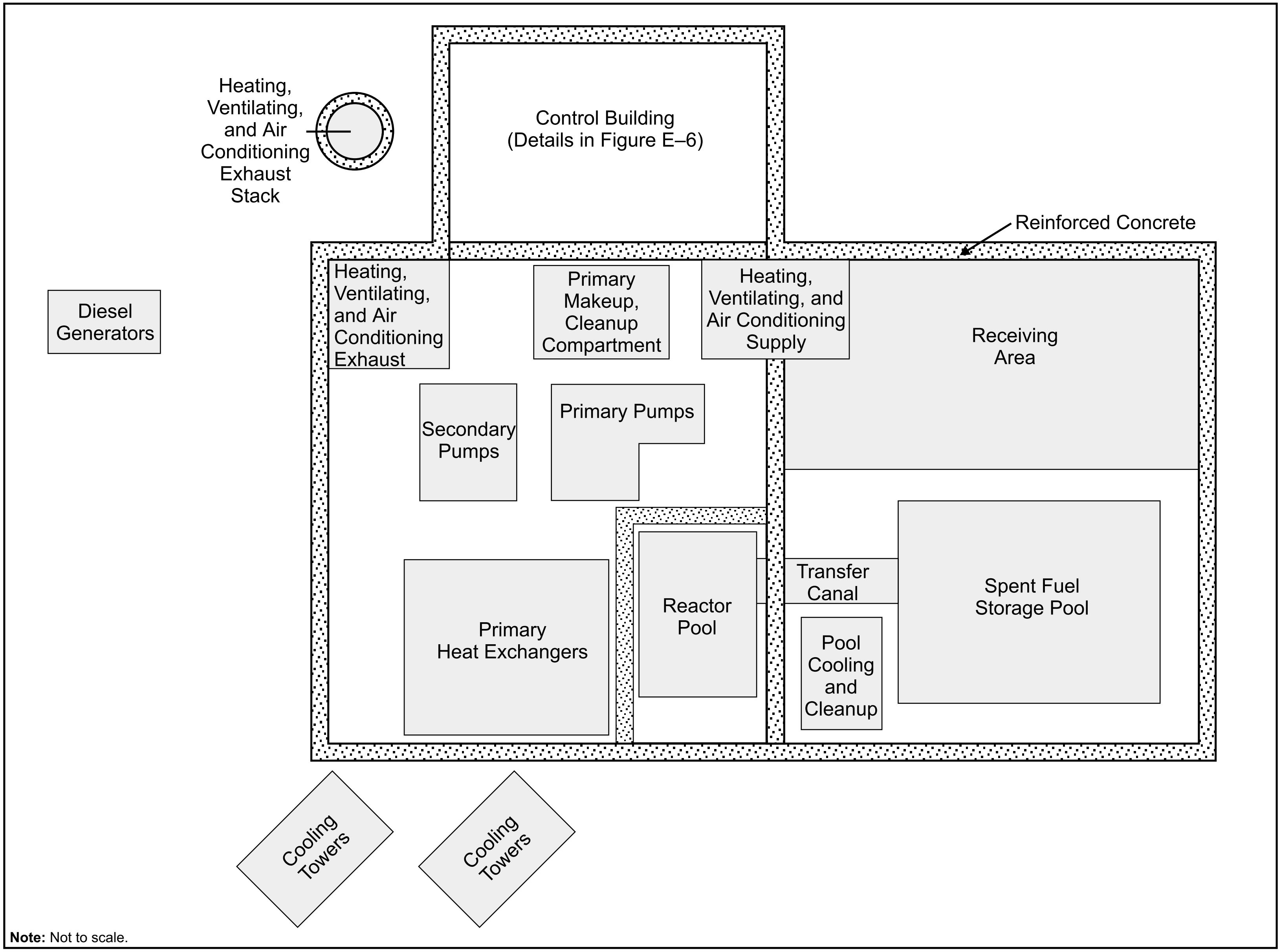

Figure E–5 Schematic of Reactor Building Complex

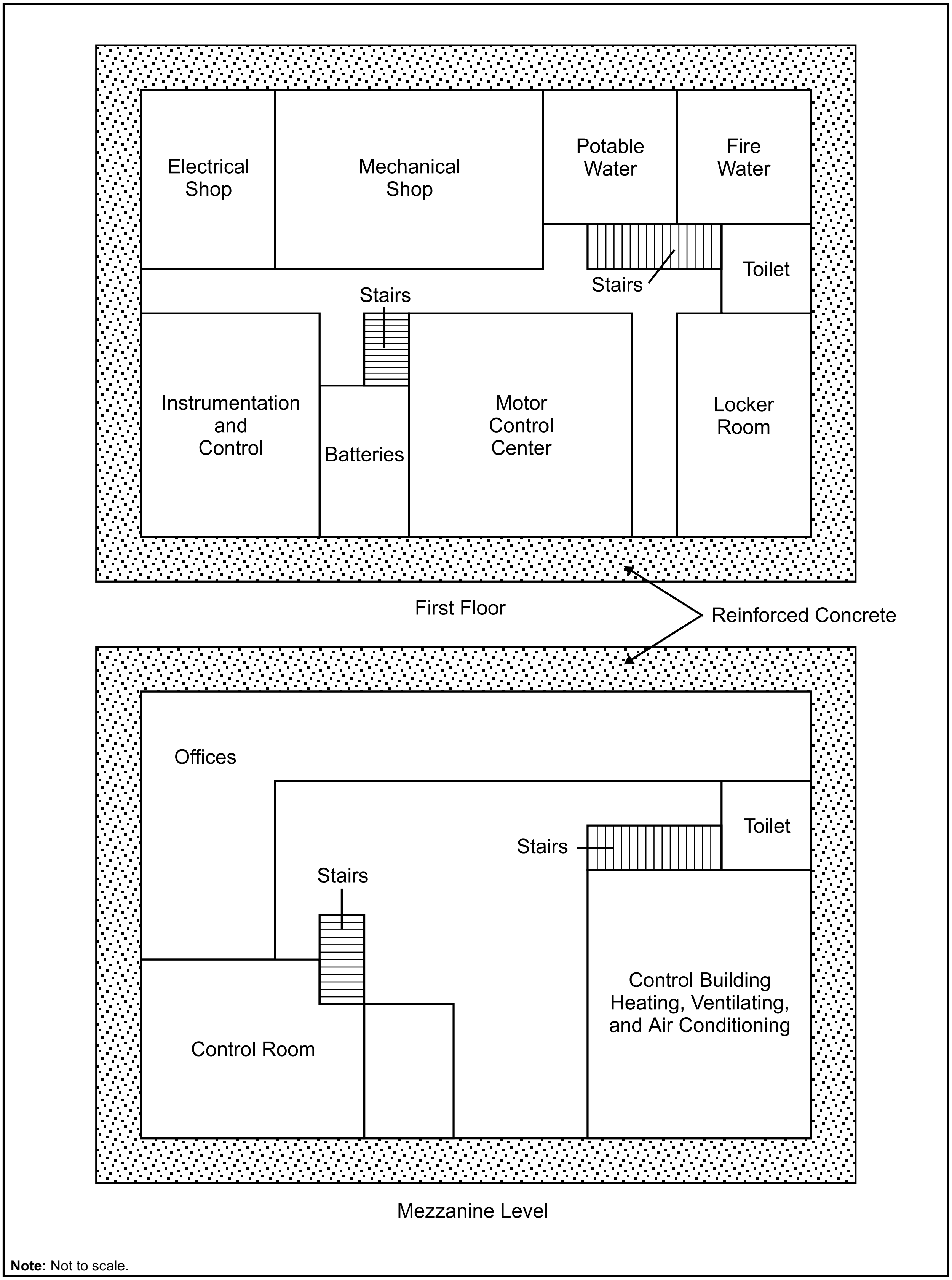

Figure E–6 Schematic of Control Building

(39,000 square feet, or about 1 acre). The total land area encompassing the facility is estimated to be approximately 4 acres.

E.8 REACTOR DESIGN SAFETY FEATURES

The new research reactor was designed with numerous inherent and passive safety features which prevent or mitigate the consequences of abnormal operational occurrences, off-normal events, and postulated accidents. The basic design constitutes an extrapolation of existing higher-power TRIGA reactors in the United States and in foreign nations. Over 6,000 TRIGA fuel elements have been fabricated and irradiated in research reactors, some with over 20 years of operation. Low-enriched uranium TRIGA fuel has been subjected to over 25,000 power pulses by the vendor with peak fuel temperatures of up to 1,150 °C (2,102 °F) without fuel damage (Simnad 1980).

The primary radiological source for this reactor is the fission products which are produced in the nuclear fuel pellets. The reactor's unique UZrH TRIGA fuel alloy has been experimentally shown to retain approximately 99.995 percent of all fission products at or below an operating temperature of 400 °C (752 °F). As presented in Table E–3, the peak and average fuel pellet centerline temperatures are less than 300 °C. The TRIGA fuel pellet also has been shown to exhibit no significant corrosion or chemical reactivity if exposed to water, steam, or air at temperatures up to 600 °C (1,112 °F). In addition, TRIGA fuel has been successfully irradiated up to a burnup of 75 percent of the available uranium-235 with no fuel damage (Simnad 1980).

The second confinement barrier for fission products is the fuel rod cladding, Incoloy-800, which can operate at much higher temperatures than aluminum and zircaloy cladding, does not oxidize at higher temperatures like zircaloy, and has a higher tensile and ultimate strength than aluminum, zircaloy, and stainless steel. Thus, the Incoloy-800 cladding provides a high degree of confinement integrity.

A third important fuel safety design feature is the inherently large negative temperature coefficient of reactivity for this fuel because it contains the hydrogen moderator intimately mixed within the solid fuel pellets. An increase in fuel temperature causes the fuel to expand, releases some of its hydrogen into the rod inner gas volume, and reduces the fuel hydrogen density. This reduction in hydrogen density shuts down the fission process and reduces power. As the fuel cools down, most of the hydrogen is reabsorbed into the fuel matrix. TRIGA fuel has always been designed to withstand sudden large power pulses and shut itself down. This pulsing feature is integral in the operation of many TRIGA research reactors worldwide.

The submerged configuration (i.e., under more than 6.1 meters [20 feet] of water) of fuel in both the reactor core and spent fuel storage pool provides another radioisotope removal mechanism if a fuel rod leak should occur. Such large depths of water absorb or retain 100 percent of all released solid fission products and over 99 percent of released halogen fission products.

The primary coolant system is designed with the following safety aspects: (1) a low system pressure (i.e., maximum less than 0.345 megapascal [50 pounds per square inch]); (2) low coolant operating temperatures (i.e., maximum hot leg temperature less than 65.6 °C [150 °F]); (3) 304 stainless steel primary coolant piping routed inside the pool to an elevation about 3 meters (10 feet) above the core before exiting the pool to the pumps; (4) a 304 stainless-steel-lined, seismically qualified reactor core pool structure; (5) primary coolant downflow through the core; (6) a long (i.e., minimum of 21 seconds) primary loop coolant transit time; and (7) a large reactor pool coolant inventory.

The low pressures and temperatures indicate that a primary coolant system pipe break would not release significant quantities of energy into the reactor building. The expected low pressures in the reactor building, coupled with the tornado missile and seismic design of this reinforced concrete structure, provide a high degree of assurance that it would maintain solid, liquid, and removable non-noble gas radioisotope confinement for all postulated accidents.

The likelihood of a primary coolant system pipe leak or break would be extremely small based on the low pressure and temperature coolant conditions and the use of 304 stainless steel that would be designed, procured, supported, and installed in accordance with all current nuclear safety standards. In addition, the elevated pipe routing above the core inside the reactor pool, in conjunction with high point vacuum breakers, would ensure that any pipe leak or break would not drain the pool below about 3 meters (10 feet) above the core. At this pool level, the remaining water would not reach saturation, with core decay heat, for at least 4 days and the water level would not boil down to the top of the core for more than 40 days. This long period for recovery of a coolant source is indicative of the large thermal heat capacity and safety margins inherent in the pool design.

As discussed earlier, the core coolant downflow direction, along with a minimum transit time of 21 seconds for the coolant to return to the pool, was selected to reduce the nitrogen-16, which is produced by neutron absorption in the oxygen component of water molecules as it passes through the core. Nitrogen-16 is a high-energy gamma-emitting radioisotope, but it has a short half-life of approximately 7 seconds. Therefore, the downflow direction and three-half-life decay delay would reduce the activity of nitrogen-16 before it returns to the pool by about a factor of 10, thereby reducing the dose rates to workers during reactor operation.

Use of the spent fuel storage pool as a source of reactor pool water is another safety enhancement of this design. The spent fuel storage pool can be hydraulically connected to the reactor pool through a transfer canal by automatically or manually opening a valve. In the extremely unlikely event that the reactor pool would lose coolant, the spent fuel storage pool is sized so that it can completely reflood the entire reactor pool volume without compromising the decay heat removal and shielding design of the spent fuel pool. Also, the higher relative elevation of the spent fuel storage pool would allow it to reflood the reactor pool by gravity-driven flow requiring no pumps. The spent fuel storage pool can be used to reflood the reactor pool without any electrical power.

The reactor core and spent fuel storage pool coolant purification system, designed to nuclear safety standards, would remove radioisotopes present in the water. In addition, the nuclear safety reactor building heating, ventilating, and air conditioning exhaust system is designed to remove 99.9 percent of airborne radioisotope particulates and 99 percent of airborne radioisotope halogens. A 36.6-meter-tall (120-foot-tall) heating, ventilating, and air conditioning exhaust stack would provide optimum atmospheric dispersion to reduce the environmental impacts of any released materials after filtration.

Along with the many natural and passive aforementioned safety features, the reactor would be equipped with a reactor instrumentation and protection system that would trip the core under specific monitored conditions including, but not limited to (1) high core power, (2) low primary coolant flow rate, (3) loss of offsite power, (4) high core exit or hot leg temperature, (5) low reactor pool level, and (6) core power distribution beyond technical specification limits. A reactor primary coolant shutdown pump is designed to circulate sufficient coolant through the heat exchanger to remove decay heat. Two 100-percent-capacity redundant emergency power diesel generators would provide adequate electrical power for all emergency systems in the event of a loss of offsite power.

The new research reactor design provides multiple layers of inherent passive and redundant nuclear-safety-related active systems to preclude, mitigate, and control any radioisotope releases to the environment and to minimize doses to both the public and workers.

E.9 REACTOR OPERATION

Operation of the new research reactor would be similar to other research reactors except that the core would be maintained at full power for a minimum of 80 percent of the year. At the beginning of a cycle of operation, neptunium-237 and long irradiation-time medical radioisotope target rod assemblies would be inserted into their appropriate fuel assembly sleeve locations. The target rods would be mechanically attached to a cluster spider assembly similar to that used for the control rod assembly. The neptunium-237 target rod assemblies would remain in the core for the entire annual fuel cycle. These target rod assemblies would be removed from the host fuel assembly without removing the fuel assembly from the core and then transferred to the spent fuel storage pool using the transfer canal.

Medical and/or industrial radioisotope target rods that require a 100-day irradiation cycle would be removed and replaced with new target rod assemblies during brief reactor shutdown periods. These target rod assemblies would be removed and transferred in a manner similar to that of the neptunium-237 target rod assemblies. Short irradiation-time radioisotopes would be inserted into rabbit tubes for the 10- to 25-day required time period. The eight rabbit tubes would be located outside the core, but inside the reflector region. The insertion and removal of irradiation targets in the rabbit tubes would have no significant effect on core reactivity and would not affect power operation.

After a radioisotope-specific cooling time in the spent fuel pool, neptunium-237, medical, and industrial radioisotope target assemblies would be transferred to a shipping cask in the spent fuel storage pool. Using the overhead crane in the spent fuel pool area, shipping casks would be placed onto a truck in the reactor building bay area adjacent to the fuel storage pool for shipment to the processing facility. New targets would be shipped from the target preparation facility into the reactor building bay by truck, transferred into the spent fuel storage pool, and subsequently moved to the reactor core pool or rabbit tube area for insertion into the core.

The plutonium-238 net annual production mission of 5 kilograms (11 pounds) was calculated to be achieved with a 300-day annual irradiation time, which corresponds to a capacity factor of approximately 80 percent. The redundant heat removal systems, low pressure and temperature conditions, and proven reactor design are expected to ensure this capacity factor. An annual shutdown for maintenance would occur during the remaining time of the year. The 10-year core refueling is not expected to affect the 80 percent annual capacity factor. Key reactor annual resource requirements are delineated in **Table E–6.**

Table E–6 Research Reactor Annual Resource Requirements

Resource Parameter	Value
Staff	120
Electricity	25,000 megawatt-hours
Reactor operating process water	7.86×10^{8} liters (2.10×10^{8} gallons)
Total water	8.07×10^{8} liters (2.13×10^{8} gallons)
Nonhazardous waste	250 cubic meters (327 cubic yards)
Hazardous waste	4 cubic meters (5.2 cubic yards)[a]
Diesel fuel	28,972 liters (7,655 gallons)
Potable and sanitary water	1.16×10^{7} liters (3.06×10^{6} gallons)

a. DOE 2000.

The annual water consumption shown in Table E–6 would be due primarily to water losses from the cooling towers which would constitute over 99 percent of the total water use. Diesel fuel consumption would be due to the monthly and annual testing of the two emergency power diesel generators. Sewer water disposal would be due to the potable and sanitary water use by the research reactor facility staff.

E.9.1 Nonradiological Emissions

During normal operations, the nonradiological emissions from the new research reactor facility would consist primarily of exhaust from testing, assumed to be a total of 72 hours per year, of the two emergency power diesel generators. These emission data were based on the estimated annual consumption of diesel fuel associated with this testing and U.S. Environmental Protection Agency (EPA) guidance on diesel engine emissions (EPA 1996). The estimated annual diesel emissions for the new research reactor are presented in **Tables E–7, E–8, E–9, and E–10**.

Table E–7 Gaseous Emission Factors and Predicted Emissions

Criteria Pollutant	Emission Factor (pounds per million British thermal units)[a]	Predicted Emissions from Subject Diesels (kilograms [pounds] per year)
Nitrogen oxides		
Uncontrolled	3.2	1,493 (3,290)
Controlled	1.9[b]	885 (1,950)
Carbon monoxide	0.85	395 (870)
Carbon dioxide	165	77,163 (170,000)
Sulfur dioxide	1.01×S[c]	472 (1,040)

a. Castaldini 1984; EPA 1979, 1996; WSPA 1990.
b. Controlled by timing ignition retard.
c. S is the percent of sulfur in the fuel, which is assumed to be 1 percent (No. 2 Diesel Fuel Oil) (Baumeister 1987).

Table E–8 Emissions Factors and Predicted Emissions for Particulate Matter

Description of Particulates	Emission Factor (pounds per million British thermal units)[a]	Predicted Emissions from Subject Diesels (kilograms [pounds] per year)
Filterable particulates[b]		
Less than 1 micron	0.0478	22.3 (49.1)
Less than 3 microns	0.0479	22.3 (49.2)
Less than 10 microns	0.0496	23.1 (51.0)
Total filterable particulates	0.0620	28.9 (63.7)
Condensable particulates	0.0077	3.6 (7.9)
Total PM_{10}[c]	0.0573	26.7 (58.9)
Total particulates[d]	0.0697	32.5 (71.6)

a. Castaldini 1984; EPA 1996.
b. Particle size is expressed as aerodynamic diameter.
c. Total PM_{10} is the sum of the filterable particulate matter with an aerodynamic diameter less than or equal to 10 microns and the condensable particulate.
d. Total particulates are the sum of the total filterable particulates and the condensable particulates.

Table E–9 Emission Factors and Predicted Emissions for Speciated Organic Compounds

Pollutant	Emission Factor (pounds per million British thermal units)[a]	Predicted Emissions from Subject Diesels (kilograms [pounds] per year)
Benzene	7.76×10^{-4}	0.36 (0.80)
Toluene	2.81×10^{-4}	0.13 (0.29)
Xylenes	1.93×10-4	0.09 (0.20)
Propylene	2.79×10^{-3}	1.30 (2.87)
Formaldehyde	7.89×10^{-5}	0.04 (0.08)
Acetaldehyde	2.52×10^{-5}	0.01 (0.03)
Acrolein	7.88×10^{-6}	0.004 (0.008)

a. EPA 1996; WSPA 1990.

Table E–10 Emission Factors and Predicted Emissions for Polyaromatic Hydrocarbons

Pollutant	Emission Factor (pounds per million British thermal units)[a]	Predicted Emissions from Subject Diesels (kilograms [pounds] per year)
Naphthalene	1.30×10^{-4}	0.060 (0.133)
Acenaphthylene	9.23×10^{-6}	0.0043 (0.0095)
Acenaphthene	4.68×10^{-6}	0.0022 (0.0048)
Fluorene	1.28×10^{-5}	0.0059 (0.013)
Phenanthrene	4.08×10^{-5}	0.019 (0.042)
Anthracene	1.23×10^{-6}	0.0006 (0.0013)
Fluoranthrene	4.03×10^{-6}	0.0019 (0.0041)
Pyrene	3.71×10^{-6}	0.0017 (0.0038)
Benz(a)anthracene	6.22×10^{-7}	0.00029 (0.00063)
Chrysene	1.53×10^{-6}	0.00073 (0.0016)
Benzo(b)fluoranthene	1.11×10^{-6}	0.00050 (0.0011)
Benzo(k)fluoranthene	Less than 2.18×10^{-7}	Less than 0.00010 (less than 0.00022)
Benzo(a)pyrene	Less than 2.57×10^{-7}	Less than 0.00012 (less than 0.00026)
Indeno (1, 2, 3-cd) pyrene	Less than 4.14×10^{-7}	Less than 0.00020 (less than 0.00043)
Dibenz (a, h) anthracene	Less than 3.46×10^{-7}	Less than 0.00016 (less than 0.00036)
Benzo (g, h, l) perylene	Less than 5.56×10^{-7}	Less than 0.00026 (less than 0.00057)
Total polyaromatic hydrocarbons	2.12×10^{-4}	0.100 (0.220)

a. EPA 1996; WSPA 1990.

E.9.2 Radiological Emissions

Radiological emissions from the new research reactor during normal operations would be due to the neutron activation of argon gas, which would be dissolved in the reactor pool water, creating argon-41 and neutron capture by oxygen atoms in water molecules, creating tritium. **Table E–11** presents the calculated annual emissions of radioisotopes from the reactor due to normal operations. Table E–11 also delineates estimated maximum annual radioactive waste emissions (DOE 2000; AECL 1996).

Table E–11 Annual Radiological Emissions Due To Normal Operation

Radioisotope	Annual Release
Argon-41	2.8 curies
Tritium (hydrogen-3)	0.1 curies
Low-level liquid radioactive waste volume	<6 cubic meters (212 cubic feet)[a]
Low-level solid radioactive waste volume	50 cubic meters (1,766 cubic feet)[b]
Transuranic waste	0
Mixed low-level radioactive waste	<0.5 cubic meter[a] (17.7 cubic feet)

a. DOE 2000.
b. AECL 1996.
Key: <, less than.

The maximum dose rate to workers at all locations within the reactor building, due to released argon-41, tritium, and direct radiation from the submerged reactor core at power or the spent fuel in the storage pool is estimated to be less than 1 millirem per hour.

E.10 Reactor Construction

Construction of the new research reactor facility was determined to require 4 years after design and licensing activities have been completed (ANSTO 1999; AECL 1996). Based on the dimensions of the reactor and control buildings, cooling towers, cooling tower separation distance from the reactor and control buildings, and ancillary structures (i.e., emergency power diesel generators, heating, ventilating, and air conditioning exhaust stack, etc.), the total surface area of structures and of the facility restricted area were calculated and are presented in **Table E–12.** This table also presents the total construction workforce; quantities of earth moved; and quantities of concrete, structural, and stainless steel estimated for the facility. It was assumed that the new research reactor would be located at an existing DOE site. Another underlying assumption is that the workforce would be present for 2,080 hours per year at the construction site.

Table E–12 Research Reactor Construction Resources

Construction Parameter	Value
Time period	4 years
Workforce	160[a]
Total reactor facility area	3,623 square meters, or 0.9 acre (39,000 square feet)
Reactor restricted area	15,942 square meters, or 3.9 acres (171,600 square feet)
Volume of earth moved	5,199 cubic meters (6,800 cubic yards)[b]
Concrete volume	5,237 cubic meters (6,850 cubic yards)[b]
Mass of structural steel	50,122 kilograms (110,500 pounds)[b]
Mass of stainless steel	3,468 kilograms (7,645 pounds)[b]
Volume of potable and sanitary water used	4.4×10^{7} liters (1.2×10^{7} gallons)
Total volume of water used	4.7×10^{7} liters (1.24×10^{7} gallons)

a. AECL 1996.
b. Tripathi 2000a, 2000b, 2000c.

During construction, pollutant emissions would be generated by vehicle operation, onsite concrete batch plant operation, wind erosion, material handling, bulldozing, scraping, and grading operations. The annual emissions for the 4-year construction period were estimated based on the workforce size, concrete production requirements, and facility areas (PFS 1997). These annual emissions are presented in **Table E–13**.

Table E–13 Annual Reactor Facility Construction Emissions

Criteria Pollutant	Annual Emissions
PM_{10} [a]	14,279 kilograms (31,414 pounds)
Nitrogen oxides	5,370 kilograms (11,815 pounds)
Carbon monoxide	6,713 kilograms (14,769 pounds)
Volatile organic compounds	1,343 kilograms (2,954 pounds)
Structural steel scrap waste	1,253 kilograms (2,757 pounds) [b]
Stainless steel scrap waste	87 kilograms (191 pounds) [b]
Concrete waste	131 cubic meters (171 cubic yards) [b]
Hazardous liquid waste	0.25 cubic meter (0.3 cubic yard)
Hazardous solid waste	0.75 cubic meter (1.0 cubic yards)
Nonhazardous liquid waste	11,400 cubic meters (14,387 cubic yards)
Nonhazardous solid waste	307,500 kilograms (676,500 pounds)

a. Total PM_{10} is the sum of the filterable particulate matter with an aerodynamic diameter less than or equal to 10 microns and the condensable particulate.
b. Tripathi 2000a, 2000b, 2000c.

E.11 DECONTAMINATION AND DECOMMISSIONING

When the new research reactor ceases its operation, it would be subject to the process of decommissioning. The reactor and its facility would be decontaminated to acceptable levels approved by the regulatory authority such that the land and buildings can be released for unrestricted uses. The research reactor and its facility then would be delicensed. For the decommissioning of a research reactor, decontamination and release for unrestricted use is generally the option chosen, although other options, such as safe storage or entombment are available for consideration.

A conceptual decommissioning plan for the proposed new research reactor emphasizes the major decontamination activities and process for the cleanup of the reactor and its facility, resulting in the final delicensing of the reactor and its facility. Part of the decommissioning plan normally includes the financial assurance requirements for the total cost of decommissioning. This requirement is expected to be exempt from the regulatory agency, since it is a DOE-owned research reactor. If DOE were to select this alternative (i.e., the construction and operation of a new research reactor at an existing DOE site), the formal site-specific decommissioning plan would be submitted for review and approval at the time of decommissioning. The decommissioning action at that time would be under a separate and appropriate National Environmental Policy Act (NEPA) review process.

E.12 References

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